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ENVIRONMENTAL AND SAFETY ASPECTS OF ALTERNATIVE NUCLEAR POWER TECHNOLOGIES - FUSION POWER SYSTEMS

OFFICE OF RADIATION PROGRAMS

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U.S. ENVIRONMENTAL PROTECTION AGENCY
OFFICE OF RADIATION PROGRAMS
LAS VEGAS FACILITY
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16. ABSTRACT

An examination of environmental and safety issues associated with advanced nuclear power technologies is being conducted by the EPA Office of Radiation Programs. Part of this effort has been devoted to a review of fusion power systems. In the past several years, progress in fusion power research and development has led to the production of conceptual power plant designs for several fusion concepts. These have included both magnetic and inertial (primarily laser fusion) confinement systems. One of the more promising concepts is the tokamak magnetic confinement scheme, and several fairly detailed power plant designs based on this concept have been produced. The tokamak designs prepared by the Princeton and Wisconsin fusion power plant design groups were used as the basis for the present review.

The present discussion includes a review of the main features of tokamak power plants with emphasis on radiological aspects. Both in-plant and environmental health physics implications of the designs are briefly reviewed. Environmental control and radioactive waste management considerations are included.

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This report has been reviewed by the Office of Radiation Programs--Las Vegas Facility, U.S. Environmental Protection Agency, and approved for publication. Mention of trade names or commercial products does not constitute endorsement or recommendation for their use.

PREFACE

The Office of Radiation Programs of the U.S. Environmental Protection Agency carries out a national program designed to evaluate population exposure to ionizing and non-ionizing radiation and to promote development of controls necessary to protect the public health and safety and assure environmental quality.

Part of this program is devoted to an examination of existing and proposed energy technologies with respect to radiological health impacts. This effort includes a review of advanced nuclear power technologies initiated during the past year. The Office of Radiation Programs--Las Vegas Facility has conducted a review of conceptual fusion power plant designs in support of this effort.

This technical note has been published in order to provide a summary to the agency, to interested professionals in other organizations, and to members of the public of health and safety aspects at an early stage in this developing technology. This report is not intended in any way to represent EPA policy with respect to any aspect of fusion power technology or its environmental and safety issues.

Readers of this report are encouraged to inform the Office of Radiation Programs--Las Vegas Facility of any omissions or errors and comments. Requests for further information are also invited.

Donald W. Hendricks
Director, Office of Radiation
Programs - Las Vegas Facility

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The author also appreciates the contributions of Daphne Prochaska and Pamela J. Pursey in typing and editorial assistance. Thanks are due to David Melton for the adaptation and preparation of the figures. Any errors in fact and interpretation are solely the responsibility of the author, however.

INTRODUCTION

Within the past five years, fusion research has moved from the realm of experimental physics into the stage where power reactor designs are being seriously discussed. A rational fusion power development program has come into being, currently funded in excess of $200 million per year. Goals are being established for the orderly development of fusion devices which are aimed at the operation of a power-producing demonstration reactor in the 1990's (Hirsch, 1975).

Recent developments include the announcement by the Energy Research and Development Administration (ERDA) of plans to build a fusion test reactor (Science, 1975a). This machine, designated the Tokamak Fusion Test Reactor (TFTR), is planned to be operational in 1981, and is expected to be the first machine capable of producing significant fusion energy. It is also the first fusion project for which an environmental impact statement (EIS) has been prepared (ERDA, 1975a).

A number of conceptual designs for large fusion power systems have been published in the United States within the past several years. Several of these are fairly detailed and include safety and environmental aspects to varying degrees. A number of other reports address general safety and environmental issues associated with fusion power. These and other cited references form the primary basis for the present discussion of fusion power systems and the associated safety and environmental issues. Even though the state of development does not permit assessments in the depth possible for current power technologies, it is possible to make some preliminary observations.

There currently exist several different major design concepts distinguished primarily by the means with which the reacting plasma or fuel is contained. For an excellent review of the major fusion reactor system concepts, the reader is referred to the article by Ribe (Ribe, 1975). The two general confinement schemes are: the use of magnetic fields to contain the plasma and the inertial confinement concept. Each of these concepts, in turn, have several variations.

Three main types of magnetic confinement systems are under investigation to achieve conditions for sustained fusion reactions in a plasma. These are: 1) low-plasma density closed toroidal systems, e.g., Stellarators and Tokamaks, 2) high-plasma density systems, e.g., linear and circular theta-pinches, and 3) open systems, e.g., magnetic mirrors (ERDA, 1975a).

A more recent, but highly-publicized concept is the inertial confinement idea. In this scheme, confinement is achieved by the compression of a solid pellet to ultra-high density at a temperature sufficient to "ignite" the constituent atoms of the fuel pellet. An explosion is achieved under high temperature and pressure conditions similar to those in a thermonuclear weapon, except that in this case, the amount of fuel is so small that only a microexplosion results from a single Deuterium-Tritium pellet. The theory is that a device could be built wherein the energy is extracted from a rapid sequence of pellet explosions in a suitably-designed chamber. Most experimental work in this area involves the use of lasers or electron beams to supply the required intense burst of energy to the pellet.

In this review, the discussion will be centered on the tokamak magnetic confinement system. This choice has been made for several reasons. The tokamak is (with the possible exception of the theta-pinch) the most thoroughly developed and documented of the available conceptual fusion power plant designs (Wisconsin, 1974; Mills, 1974; Fraas, 1975). This concept receives about 60 percent of the total Federal government support for magnetic confinement approaches currently under development. It is the basis for the first large fusion test reactor facility (the TFTR).

The laser fusion concept, while highly publicized as a future power source, is being pursued in this country primarily as a means of simulating nuclear weapons explosions (Science, 1975b). In addition, the published laser-fusion conceptual power plant designs do not contain sufficient detail to support more than a limited comparison of environmental considerations with other nuclear power systems.

THE TOKAMAK CONCEPT

Controlled thermonuclear fusion devices whose region of reacting plasma is contained in a closed toroidal, or doughnut, configuration are now generally referred to as tokamaks. This is the name of a Russian device of toroidal configuration which in 1968 achieved a major increase in plasma confinement time and ion temperature (ERDA, 1975a).

The name for that machine is an acronym derived from the Russian words for Toroidal Chamber Machine (ERDA, 1975a). An innovation which distinguished the tokamak from other toroidal plasma devices was the use of an induced electric current in the plasma itself. This simultaneously provides additional plasma heating and generates a secondary magnetic field around the minor circumference of the torus (poloidal field) which is necessary for increased plasma confinement time in the toroid. Figure 1 illustrates the toroidal geometry and Figure 2, the main features of a tokamak confinement system.

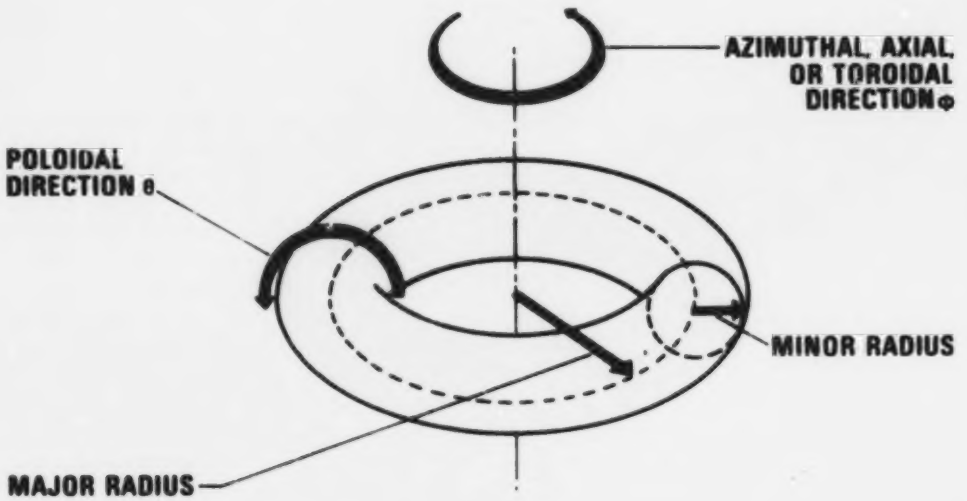

Figure 1. Toroid Geometry

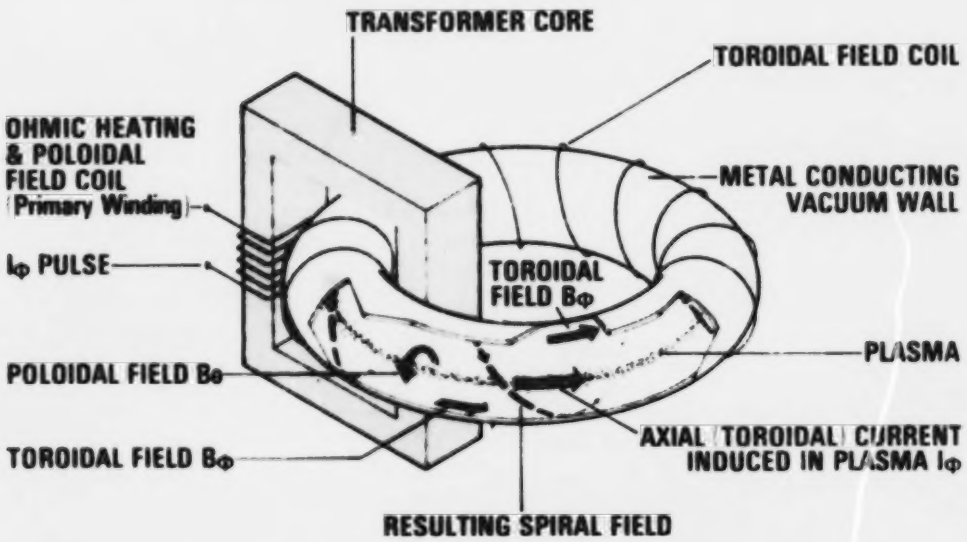

Figure 2. Main Features of a Tokamak

In the tokamak, the plasma itself forms the secondary of a set of transformer cores. In this case, the secondary is highly-ionized (hence, electrically conducting) gas instead of a solid conductor, as in a conventional transformer. A strong current pulse in the primary windings ionizes the gas and generates an axial (or toroidal) current in the plasma, as shown in Figure 2. This produces two effects: 1) heating of the plasma, and 2) generation of a magnetic field, called the poloidal field, perpendicular to the direction of the axial plasma current. A second magnetic field is generated in the axial direction (parallel to the plasma current) by the toroidal field coils. The resulting field lines from the combination of the toroidal (axial) and poloidal fields form a helical configuration as indicated by the heavy dashed line in Figure 2. Actually, due to the large number of lines formed in this manner, the resulting field creates a confining structure which resembles the crossed plies of an automobile tire casing, as described by Rose (Rose, 1971).

These features comprise the essential ingredients of a Tokamak confinement system; however, a number of additional requirements must be met in order to achieve energy production. In a fusion power system, conditions of plasma density, temperature (average ion kinetic energy), and confinement time, sufficient to sustain the reaction between the plasma components, must be achieved and maintained. The energy produced in the reacting plasma must be extracted, the reaction product atoms (ash) removed, and the plasma replenished, or fueled, on a continuous basis. Cooling is required and the large plasma chamber must be maintained at a relative vacuum.

HISTORY OF FUSION RESEARCH

The fusion reaction was discovered in the early 1930's and was identified as the mechanism by which the sun and most stars produce energy. Achievement of net energy production by fusion first occurred via the thermonuclear weapon. The first such explosion took place in 1952. At about this time, research was initiated in this country, and elsewhere, on controlled fusion, as the possible advantages of such a means of producing power were realized. A number of fusion reactions are possible, including the Deuterium-Tritium or D-T reaction, which can be fueled by deuterium, for which an essentially inexhaustible supply exists: However, it was also recognized that no known or conceivable solid material could be used to contain the reaction at the 100 million degree Kelvin temperature necessary for its sustainment (ERDA, 1975a). Hence, early investigations centered on the use of magnetic fields to confine the reacting fuel of hot plasma.

As knowledge was gained about plasma behavior, particularly the mechanisms by which plasmas tended to dissipate, early optimism for achievement of controlled fusion was replaced by efforts aimed

at gaining necessary further understanding of plasma behavior. Today, some forty years after the discovery of the principle, sustained, controlled fusion has not yet been achieved. In contrast, four years elapsed between the discovery of nuclear fission and the first sustained fission experiment. Fusion research has been, in general, characterized by a series of small successes in achievement of conditions leading toward a sustained, controlled fusion reaction. This has, by no means, been an orderly progression, as a number of sometimes unexpected setbacks have been experienced and the solutions sought.

STATE OF DEVELOPMENT

State of the Technology

The current status of fusion power research and development can be discussed by first examining, quantitatively, the parameters that describe the conditions required for the production of net energy in a plasma. The basic requirements are that a reacting plasma of sufficient density must be confined at high temperature for a time which allows enough reactions to occur for net energy production. In 1957, the British physicist, J. D. Lawson, developed a simple relationship for the energy balance in a plasma volume (ERDA, 1975a). Under the assumption that the energy from the reactions can be recovered at about 33 percent efficiency, the product of the plasma density and confinement time must be at least 10^{14} sec/cm^3 at an average plasma temperature of about 10 KeV.* This is the so-called Lawson criterion for minimum conditions required for a successful fusion reactor (Tuck, 1971).

To date, no single device has achieved the simultaneous conditions of temperature, density, and confinement times necessary for net energy production. A number of machines of different types have, in recent years, shown a rather steady advance toward the region of success. For example, some specialized devices have exceeded the fusion ignition temperature in a plasma for sub-second durations. Others, such as tokamaks and their near relatives, have pushed the density-confinement time product to about 10^{13} sec/cm^3, but at temperatures of about 1 KeV as opposed to the 10 KeV necessary for ignition of the D-T reaction (Nucleonics Week, 1975).

A statement by R. L. Hirsch, of the U.S. ERDA, provides a capsule summary of the present state of development (Hirsch, 1975):

> Practical generation of fusion energy requires that the fuel plasma be held at a specified

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* The value of 10^{14} sec/cm^3 is for the Deuterium-Tritium, or D-T reaction, which is the most energetically-favorable fusion reaction. The value would be higher for other reactions; for example, about 10^{16} sec/cm^3 for the D-D reaction (Chen, 1974).

density and temperature for sufficient time to result in the desired energy release. In magnetic confinement fusion research, minimum practical plasma densities were first realized in 1953, minimum temperatures in 1962, and adequate confinement in 1969. It is necessary that all three of these conditions be achieved at one time in one experiment in order to have a meaningful fusion rate take place. At present only combinations of two can be achieved, because sufficiently large magnetic "bottles" have yet to be built.

Table 1 summarizes the current status of key parameters achieved in tokamaks and compares them with minimum requirements for energy break-even. In addition, values are shown for typical power plant conceptual designs. There are currently at least 15 tokamaks in operation and seven under construction in various countries (Prevot, 1975). Present devices are fairly small (with major radii on the order of two meters or less), operate with conventional magnets, and use hydrogen or deuterium plasmas. Devices currently in the design and planning stage include: scaling to larger sizes, the use of super-conducting magnets, and the introduction of deuterium-tritium (D-T) plasmas.

In summary, fusion experiments are moving from an emphasis on plasma physics into the state where engineering development and technology problems are under active investigation.

Limitations to Development

A number of technical areas have been identified in which further development is required or in which present problems must be overcome before large power systems will be feasible. These are separate from the demonstration of net energy production in a plasma experiment. The following discussion summarizes the major problems and is centered on tokamak systems.

a. Plasma Engineering

The temperatures attainable via plasma current ohmic heating and reaction product absorption are not sufficient to sustain the fusion reactions.* Means of supplying the additional energy are being investigated. A promising method is through the injection of energetic neutral particles, i.e., neutral beams. Other plasma heating methods under investigation include: injection of relativistic electrons, radiofrequency and microwave heating, and heating by adiabatic compression of the plasma.

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* Alpha particles produced along with neutrons in D-T reactions supply energy to the plasma as their range is on the order of plasma dimensions.

TABLE 1. PLASMA PARAMETERS IN TOKAMAKS

	CURRENTLY ACHIEVED	MINIMUM FOR ENERGY BREAK-EVEN	TYPICAL FOR LARGE POWER PLANT CONCEPTUAL DESIGNS (RIBE, 1975)
Plasma Density (elec/cm^3)	6×10^{14} (a)	10^{14}	10^{14} to 10^{15}
Ion Temp. (KeV)	1.6 (b)	5 to 10	20 to 40
Particle Confinement Time (sec)	0.1 (c)	.01 to 1	10

(a) Mass. Institute of Technology ALCATOR experiment (Hirsch, 1976).

(b) French Tokamak Experiment (TFR)(Kinter, 1976).

(c) Soviet Union T-10 tokamak (Nucleonics Week, 1976).

Another problem is the necessity for maintaining plasma "purity", i.e., minimizing the number of foreign, parasitic, non-reacting ions in the plasma. The impurities include reaction product alpha particles (He^4 ions) and ions knocked off the vacuum chamber walls by interactions with energetic plasma particles. Fuel ions will have to be continuously supplied to the plasma in order to maintain the reaction rate. Some tokamak designs provide for this through neutral particle beam injection, while others propose to inject small D-T solid pellets, or provide a fuel gas blanket around the burning plasma to simultaneously fuel the plasma and minimize impurities.

It also remains to be demonstrated that plasmas can be scaled to larger volumes to attain the power levels appropriate for reactors, as predicted by theory, without introducing destructive instabilities (Pease, 1975).

b. Tritium Technology

Tritium inventories in large fusion power systems are expected to be on the order of tens of kilograms. At approximately 10^7 Ci per kg, this represents amounts of tritium much greater than ever handled previously. In addition to the control requirements imposed by such large inventories, methods need to be developed for tritium recovery from the plasma exhaust, and from the coolant-breeding material (Darvas, 1975). In contrast to the large total inventory of tritium in a reactor system, the tritium in a breeding blanket, e.g., molten lithium, will be very dilute, on the order of parts per million (Ribe, 1975). Tritium is very mobile in most environments; diffusion rates through structural materials and system boundaries present problems.

c. Material Radiation Damage

Large regions of a D-T fusion reactor will be exposed to heavy fluxes of 14 MeV neutrons. This includes structural materials, the vacuum wall, and the coolant system. Radiation damage to the vacuum wall and surrounding materials may require that they be replaced during the reactor lifetime, or that the wall be protected from the full fluence by a neutronically-inert shield such as carbon.

Little is currently known of the behavior of candidate materials in the environment inside a fusion reactor (Darvas, 1975). Sources of 14 MeV neutrons in the intensity range of interest are currently not available for materials investigations.

d. Blanket Design

Referring to Figure 3, the blanket generally includes the region bounded by the vacuum wall and the magnetic field windings in a typical tokamak conceptual design. The blanket serves several

functions. It provides for neutron moderation and energy removal. Tritium is generated by neutron reactions in a suitable material, e.g., Li-6, which is generally a constituent of the main coolant. The blanket region must also shield the magnetic field coils from the neutron and gamma ray fluxes. Cooling must be provided for the vacuum or first wall, which must be designed to withstand severe thermal and radiation fluxes in addition to its vacuum barrier function.

In most conceptual designs, the main coolant is a liquid metal such as a lithium compound. This means that an electrically-conducting fluid must be pumped through the strong magnetic field surrounding the reactor. Considerable power must be available for pumping the coolant and cooling passages must be designed to minimize cross-field flow (Lidsky, 1972). The understanding of turbulent liquid-metal heat transfer in the presence of strong magnetic fields is important for blanket design. However, the current understanding is incomplete due to the lack of appropriate experimental data (Lykoudis, 1975). The use of a gaseous coolant would eliminate the problems associated with liquid metal coolants. This approach however, would increase the difficulty in attaining a good tritium-breeding ratio which is associated with liquid lithium.

The problems of blanket design are severe; a number of designs incorporating a variety of materials have been examined.

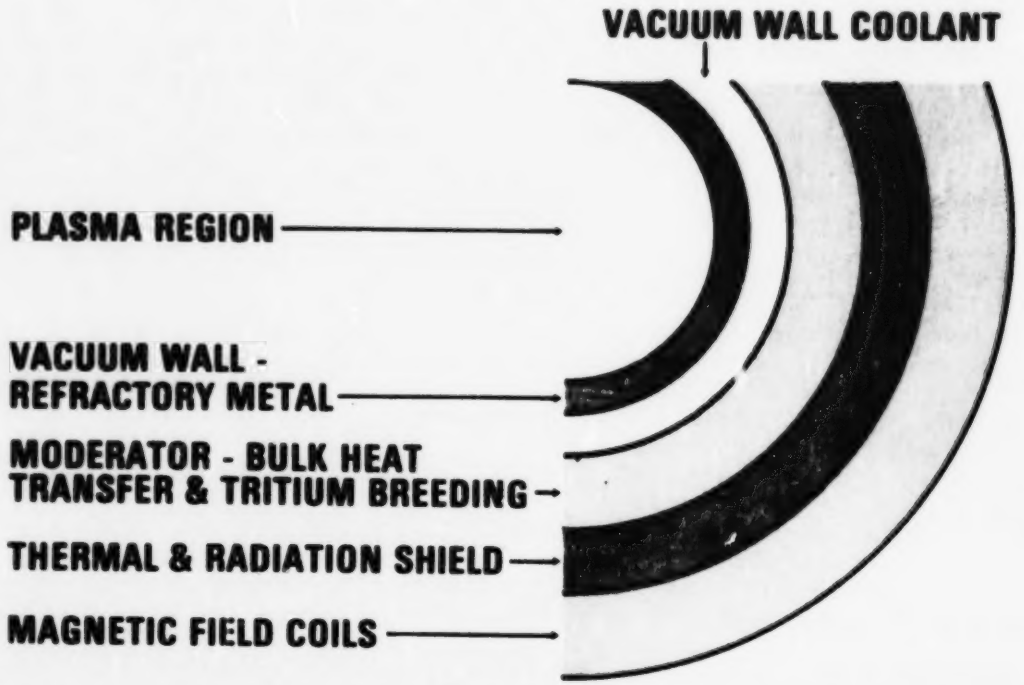

Figure 3. Schematic of Tokamak Fusion Reactor - Blanket Zones

Many of the unresolved questions must await the construction and operation of actual test reactors.

e. Magnet Technology

Attainment of the necessary magnetic field strengths for fusion reactors is feasible through the use of super-conducting magnets. However, there is no experience in constructing, handling, and operating magnets of the size required for a fusion reactor with a plasma chamber several meters in diameter (Darvas, 1975). The structural requirements to withstand the stresses generated by the magnetic forces are imposing.

f. Fast Electromagnetic Energy Storage

Several tokamak reactor designs operate in the pulsed mode. This requires the ability to shift large amounts of energy in very short times, e.g., on the order of seconds, into magnetic field windings. This precludes the use of line power directly, hence rapid inertial or electromagnetic storage systems must be used. Such systems need to be developed in order for large pulsed systems to be operable (Darvas, 1975).

In addition to the technical problem areas summarized above, economic and institutional constraints will control the implementation of fusion power systems into the national energy grid.

An overall constraint on the development necessary to demonstrate the feasibility of large fusion power systems is cost. If the technical problems are overcome, but capital or operating costs of the resulting power systems are not competitive with other energy technologies, the designs will be unacceptable. Successful implementation of fusion power will require the development of an industrial base to produce fusion power systems as the technical development problems are surmounted (AEC, 1973).

Research and Development Support

The support for fusion power research and development currently comes primarily from the Federal government, administered by the ERDA. During the fiscal years 1951 through 1974, the ERDA predecessor, the Atomic Energy Commission (AEC), spent $544 million on magnetic confinement research (GAO, 1975). Table 2 summarizes magnetic confinement funding and projections for the fiscal years 1975 through 1980. An additional $116 million has been spent by the AEC on laser fusion research and development in the period, 1963-1974 (GAO, 1975). The 1975-1980 projections for laser fusion funding are about one-third of the ERDA funding for magnetic confinement research and development.

The major ERDA-funded magnetic confinement research and development efforts are carried out primarily at four laboratories.

TABLE 2. ERDA-CONTROLLED THERMONUCLEAR RESEARCH FUNDING PROJECTIONS 1975-1981 (MAGNETIC CONFINEMENT PROGRAM ONLY--DOES NOT INCLUDE LASER FUSION, DOLLARS IN MILLIONS) †

FY-75	FY-76	Transition Period	FY-77	FY-78	FY-79	FY-80	FY-81
105*	192**	62	360	471	567	559	634

† From ERDA, "Fusion Power Program Research and Development Program Projections, May 1975, draft.

* Excludes $6 M deferral.

**Estimate

These are: Lawrence Livermore Laboratory (LLL), Livermore, California; Los Alamos Scientific Laboratory (LASL), Los Alamos, New Mexico; Oak Ridge National Laboratory (ORNL), Oak Ridge, Tennessee; and the Princeton Plasma Physics Laboratory (PPPL), Princeton, New Jersey. The remainder of the program is carried out at a number of private firms and universities. About 75 percent of the current magentic confinement research and development program is conducted at the four ERDA laboratories. Additional fusion research and development support is provided by private concerns, electric utilities (partially through consortia), and the Electric Power Research Institute (EPRI).

The General Atomic Company has supported a fusion reactor development program for a number of years and is now receiving major ERDA support for the design and construction of a large non-circular cross section tokamak experiment. Westinghouse Electric Company participates in the PPPL program and United Aircraft Corporation supports some in-house fusion research and development. Estimates of the amount of support for magnetic confinement fusion research and development from sources other than ERDA are difficult to obtain. In 1973, it was estimated that other governmental agencies supported various efforts at the level of about $2 million per year, and that the total U.S. private-industrial effort was in the range of $6 - $7 million (AEC, 1973). A recent estimate of the total U.S. non-ERDA spending, including other government and private sources, is between $15 - $20 million per year for the next five years (Katz, 1975).

With respect to foreign fusion research and development programs, the following summary is included (AEC, 1973):

> Abroad, there are major fusion programs in Europe, the Soviet Union, and Japan. The European program is now unifying under EURATOM and is about twice the size of that in the U.S. The Japanese effort is possibly 20-30% of the U.S. program, while that in the Soviet Union is between two and three times larger. The Soviet effort encompasses the same features as that in the U.S., with emphasis on the Soviet-developed Tokamak concept, the Stellarator, and laser fusion. The USSR has about 50 major CTR and plasma research experiments in operation at seven major research centers; the Soviets have obtained results on a par with, and in many cases have surpassed, the U.S. program. In addition, the Soviets have an outstanding theory program. In magnetic confinement research there has been excellent international cooperation for many years. In laser-fusion there is good communication on portions of the problems, but classification precludes discussion in other parts.

Finally, the ERDA-DCTR objectives and strategy for fusion power development, as recently announced, are given (ERDA, 1975b):

Objectives

Near-Term (-1985): Produce reactor level hydrogen plasmas. Produce substantial quantities of thermal energy in the First Fusion Test Reactor using Deuterium-Tritium fuel.

Mid-Term (-2000): Produce electrical energy in substantial quantities in two Experimental Power Reactors between 1985 and 1990. Operate commercial scale Demonstration Power Reactor (1997).

Long-Term (+2000): Begin supplying a fraction of the Nation's electrical energy demand.

Strategy

A series of progressively larger experimental devices will provide needed knowledge of fusion plasma physics and engineering under prototypical fusion reactor conditions. This will permit an evaluation of the different types of fusion systems and serve as the basis for the design and operation of fusion power reactors.

A combination of industrial, academic, and National laboratory resources will be used with funding support from EPRI and utility consortiums, where possible, to expand the scope, hasten the pace, and prepare the technology for full commercialization.

All subprogram efforts are interrelated with the goal of supporting a successful fusion demonstration by the most promising concept (currently the tokamak) and also to continue to develop feasible alternatives as backup.

POTENTIAL SIGNIFICANCE TO ENERGY SOURCES

The main attraction of fusion power is that the fuel supply is essentially inexhaustible. Deuterium occurs in sea water in a ratio of one to every 6,500 hydrogen atoms (Lidsky, 1972). Deuterium separation is relatively easy and inexpensive. Tritium is produced by neutron capture reactions in lithium, of which there appears to be an adequate supply for a full-scale fusion power economy (Braams, 1975).

Deuterium fuel requirements are on the order of two kilograms per day for the reference designs and can be obtained for about $60 per kilogram (Mills, 1974). The cost of fuel for fusion power plants is so low per unit of energy produced that fuel costs are removed as a consideration in fusion power potential assessments (Hafele, 1974). These fuel materials are not expected to be in demand for other energy technologies.

Resource requirements for power plant construction depend, to a certain extent, on the particular materials used in chosen designs. In addition to conventional material requirements, such as steel and concrete, which are expected to be similar to conventional light water fission reactor (LWR) and fossil plants in the balance of a plant, a full-scale fusion economy could pose severe demands on some material resources. For example, He, Be, Nb, Cr, Mn, and Sn are relatively scarce, and some designs require the use of large amounts of these materials (Wisconsin, 1975).

TECHNICAL DESCRIPTION OF CONCEPTUAL TOKAMAK POWER REACTOR

The details of large fusion power systems will be drawn from the recent Wisconsin and Princeton designs. These represent the most complete and best documented of the tokamak power system designs currently available.* Referring to the schematic in Figure 4, the main features of a conceptual tokamak power system are presented. It is convenient to discuss the power plant features in terms of the following systems: plasma, divertors, fuel system, blanket and shield, magnets, cooling system, and steam generators.

Each system will be described in terms of its purpose and some design parameters will be presented. Where appropriate, comparisons to LWR design parameters will be made. In many instances, such comparisons will be of limited value, due to the inherent differences in fission and fusion physics and the associated design requirements for each type of power system. It must be acknowledged that the present discussion is limited to a very cursory examination. Further, the design reports under consideration, in many cases, do not include details of design and operation of all the systems associated with a power plant. This is to be expected in conceptual designs which, in many cases, can only discuss alternatives or possible ranges of design parameters.

System features and operating characteristics for the two reference designs are presented in Table 3. Both machines are deuterium-tritium fueled and breed tritium from lithium contained in the blankets. The main toroidal fields are produced by superconducting magnets in both designs. The UWMAK main coolant is lithium with an intermediate sodium loop driving the steam generators. The Princeton machine primary coolant is helium, which is the primary of a specially-designed, helium-heated steam generator. Both designs use conventional steam turbine plants to generate electricity.

The duty cycles of both machines are similar in that they are designed to operate on a long pulse of 90 to 100 minutes with a recovery time of several minutes until plasma reignition occurs.

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* It should be noted that these designs represent very early attempts to describe systems that may be built 20-30 years in the future. Actual design configurations can be expected to be considerably different, as the technology progresses and new alternatives are evaluated.

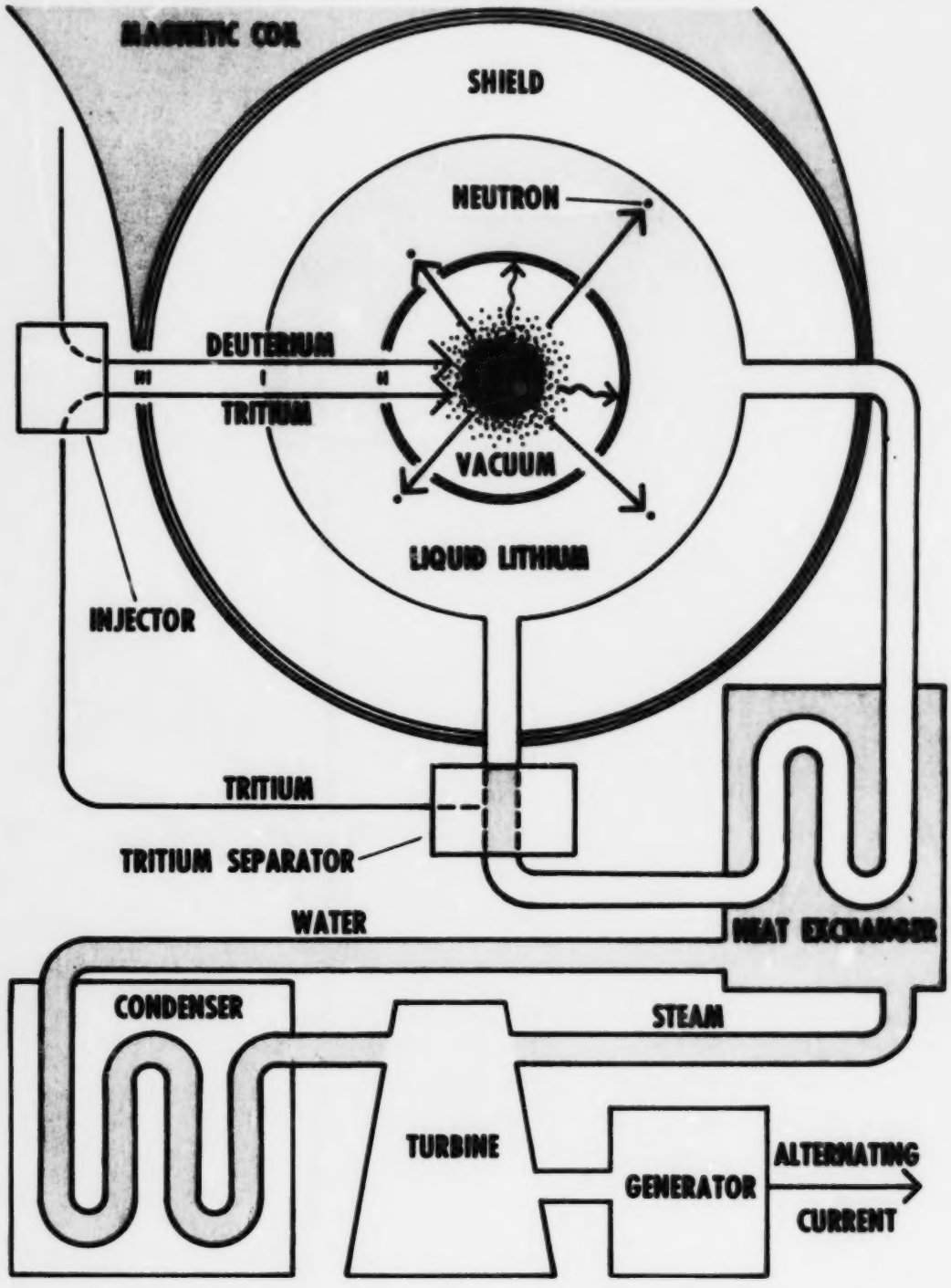

Figure 4. Tokamak Fusion Power Plant Schematic

TABLE 3. TOKAMAK REACTOR REFERENCE DESIGN FEATURES

	UWMAK-I	PPPL
Power: MWt	4,660	5,305
MWe (net)	1,473	2,030
Net Electrical Efficiency (%)	32	38
Fuel Cycle	(D-T) Li	(D-T) Li
Toroidal Vacuum Chamber:		
Major Radius (meters)	13	10.5
Minor Radius (meters)	5	3.25
Vacuum - First Wall Structural Material	316 - stainless steel	PE-16 - austenitic nickel alloy
Shielding Materials	Pb, SS, B_4C	high-density concrete, Pb, B, polyethelene
Main Coolant	Lithium	Helium
Moderator--Tritium-Breeding Material	Lithium	$(LiF)_2BeF_2$ eutectic mixture (flibe)
Magnets: Super-Conducting Material	Nb Ti + Cu	Nb_3Sn
Power Cycle	Li-Na-Steam	He-Steam
Plasma Burn Cycle:		
Burn Time (min)	90	100
Recovery Time (min)	6.5	3
Thermal Flywheel	Sodium storage loop	flibe diverted from blanket

In order to provide continuous electric power output, thermal flywheels are incorporated. During the plasma recovery phase, energy to the steam generators is provided by a special reservoir of coolant.

REACTOR SYSTEMS

Plasma and Burn Cycle

Reference design plasma characteristics are summarized in Table 4. Operating cycles of these machines are characterized by long pulses, i.e., the burn time is much longer than the residence time of the plasma particles. Consequently, fuel ions must be continuously supplied, and the reaction products and unburnt fuel removed to maintain plasma conditions during the burn cycle. Each machine accomplishes plasma management and burn cycle control in a slightly different manner. The Wisconsin Tokamak burn cycle utilizes neutral deuterium and tritium beam injection following plasma current (ohmic) heating to achieve plasma ignition conditions; whereas, in the Princeton design, ignition is achieved by ohmic heating and D-T ice pellet injection. During the burn phase, D-T pellet injection is used to maintain the plasma in both machines.

After the burn phase is terminated, due to a number of factors including resistive dissipation of the plasma current and build-up of plasma impurities, the current is shut down, and the plasma chamber evacuated. Burn cycle times are given in Table 3. During the recovery phase, the field coils are re-energized, fresh D-T gas is injected, and the cycle is re-initiated.

Divertors

As stated by Ribe (Ribe, 1975):

> The basic objective of a divertor is to prevent particles diffusing out of the plasma from hitting the first wall and to provide a means of removing alpha particle "ash" and impurities, while maintaining a steady throughput of fuel from injectors. The divertor also provides a shield for the hot plasma against impurities released from the wall, or flowing back from the divertor chamber.

The divertors consist of a system of specially-shaped secondary magnetic fields at the plasma periphery, baffles and flow channels, and vacuum pumps and manifolds. Particles which diffuse outward from the active plasma zone through the main confinement magnetic field are channeled by the divertor poloidal field to particle collectors. Vacuum-pumping systems sweep the diverted materials to gas cleanup systems for tritium separation and gas disposal.

TABLE 4. REFERENCE DESIGN PLASMA CHARACTERISTICS

	UWMAK-I*	PPPL
D + T Ion Density (cm^{-3})	0.8×10^{14}	10^{14}
Ion Temperature (KeV)	11	30
Average Ion Confinement Time (sec)	14.2	3.8
Plasma Volume (m^3)	6,400	2,190
Plasma Current (Amp)	21×10^6	14.6×10^6
Plasma Power Density (MW/m^3)	0.8	2.4

* UWMAK plasma parameters are given for operation at thermally-unstable equilibrium.

The divertor vacuum system also purges the main plasma chamber at the end of each burn cycle. The Princeton design uses two toroidal vacuum manifolds connected to the inner circumference of the main plasma chamber. The vacuum and pumping requirements are met by 96 mercury diffusion pumps. The UWMAK divertor-vacuum system also uses 96 mercury diffusion pumps with freon and liquid, nitrogen-cooled traps. Two sets of two divertor slots provide access to the main plasma vacuum chamber.

Vacuum requirements for the fusion power plants are well within the capabilities of current vacuum technology, i.e., pressures down to 10^{-8} torr (1 torr = 1 mm Hg). Design challenges are posed by the exceedingly large vacuum chamber volumes and large pumping speeds required. Pumping speeds within the range of multi-million liters per second are required to purge the plasma chambers at the end of each burn cycle.

Fuel System

Fuel ions are supplied to the plasma in several ways. At the beginning of the burn cycle, deuterium and tritium gas is pumped into the plasma chamber. After the initial fuel charge is ionized, heated, and ignited, it is proposed to supply makeup fuel ions either by injection of neutral beams or solid pellets. During the plasma burn phase, both conceptual designs under consideration accomplish fuel supply by solid D-T pellet injection. During the ignition phase, the Wisconsin design uses neutral beam injection, but the primary purpose here is to supply energy to the plasma.

As the plasma burns, deuterium and tritium ions are lost by fusion reactions and diffusion out of the active plasma region. The combined loss rate for deuterium and tritium ions in these designs is about 3 to 4 x 10^{6} ions/second. To maintain the plasma, fuel ions must therefore be injected at this rate. The UWMAK design proposes to inject D-T pellets of 20 micron radius into the plasma by electrostatic accelerators, at the rate of 20 x 10^{6} pellets per second.

Tritium is recovered from the plasma exhaust and the breeding blanket for reinjection into the plasma. Principal fuel flows for the Princeton design are shown in Figure 5. In this design, cryogenic fractional distillation is used for gas separation in the fuel exhaust purification system. The Wisconsin design plasma exhaust system traps the gases on the lithium divertor coolant and divertor vacuum pump traps. The design proposes to use yttrium extraction beds to remove tritium from the divertor coolant and cryogenically-cooled charcoal traps for tritium removal at the pump traps.

Table 5 gives typical material flow rates in the Princeton primary fuel loop. The argon is used as the source of impurity ions which are injected into the plasma for burn control and quenching at the end of the cycle.

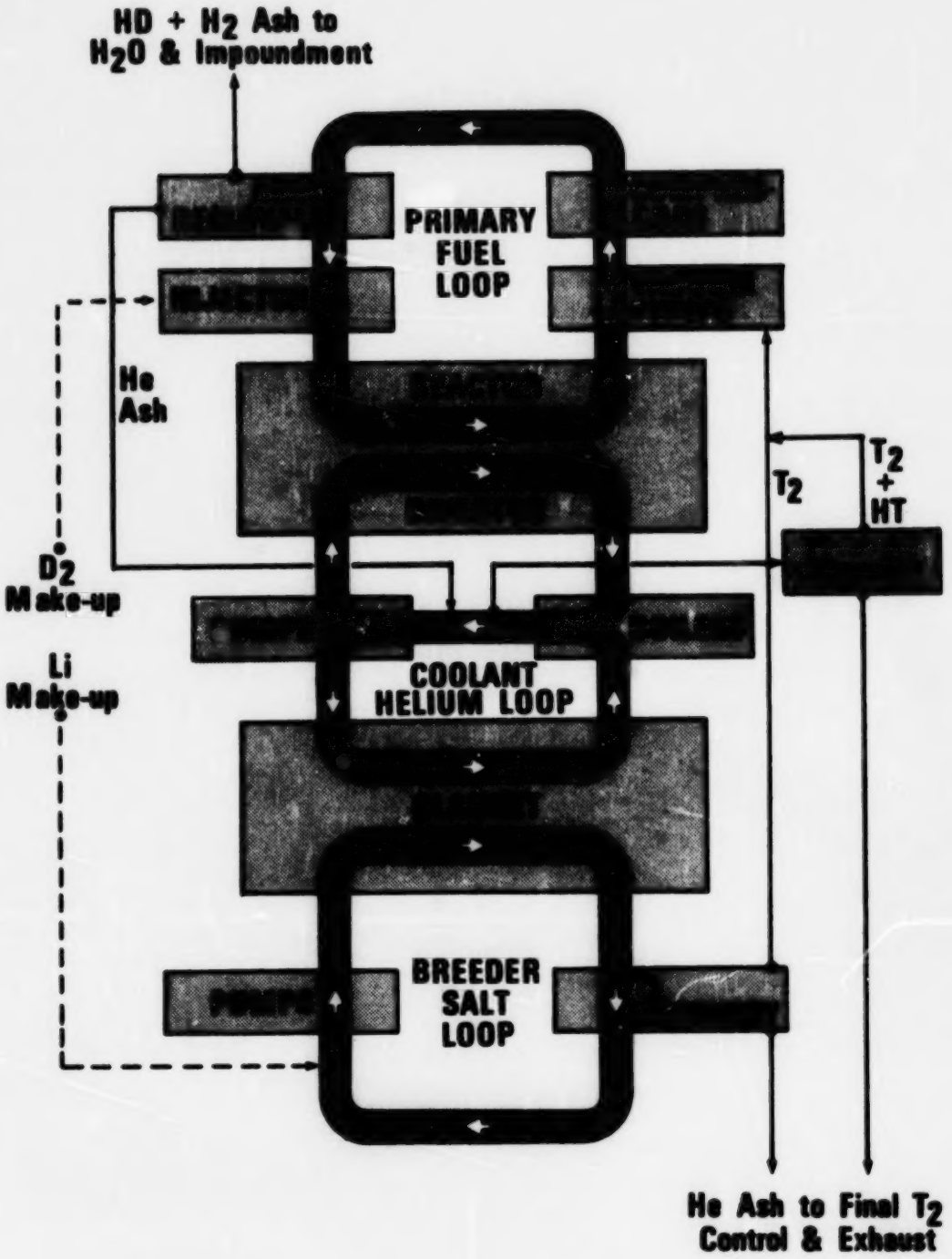

Figure 5. Principal Fuel Flows in Princeton Tokamak

TABLE 5. FLOW RATES IN PRINCETON TOKAMAK PRIMARY FUEL LOOP (GRAMS/HR)(MILLS, 1974)

COMPONENT	REACTOR FEED	VACUUM PUMP EFFLUENT
He		29.3
H		0
HD		0.15
HT		0.10
D_2	77.1	77.0
DT	219.8	182.8
T_2	0	21.5
Ar	252.8	252.8

Blanket and Shield

Blanket design requirements are summarized in the Wisconsin design report as (Wisconsin, 1974):

-- Tritium breeding.

-- Conversion of D-T reaction product kinetic energy into heat.

-- Conduction of the heat to an electrical generation system.

-- Provision of: vacuum-tight enclosure for the plasma; passages for fueling, heating, and diagnostic equipment; and coolant channels to conduct the heat away.

Details of blanket and shield for the PPPL reactor are shown in Figure 6. Due to radiation damage, both reactor designs will require replacement of the first, or vacuum wall, during the reactor operating life. The design goal for the Princeton first wall is five years and the expected life in the Wisconsin design is two to three years. Both reactors use a blanket made up of modular segments, like the sections of an orange, in their placement around the torus. The PPPL blanket consists of 24, and the UWMAK of 12, identical segments.

"Each module consists of a super-conducting toroidal magnet, blanket and shield segment, vacuum pumps, coolant headers, and all the other necessary plumbing and electrical connections for fueling and heating the plasma" (Wisconsin, 1974). The modular design is intended to provide ease of blanket systems maintenance by allowing the removal and replacement of an entire module, which can be repaired and maintained while the reactor operates with the replacement module.

Materials and coolants for the two blanket designs are given in Table 3. Shielding is incorporated into the blanket for several purposes: to provide protection to blanket materials from thermal loads; to protect materials and systems from radiation damage; and finally, to provide biological protection from the intense neutron, gamma, and x-ray fields generated in the reactor. In certain areas, cryogenic magnets are located close to the plasma, e.g., stabilizer and divertor coils. These systems in particular require protection from thermal loading and radiation damage. Most of the shielding is designed with provision for internal cooling.

Magnets

Magnet characteristics for the two designs are presented in Table 6. Both reactors use super-conducting, helium-cooled magnets for all major magnet systems. As indicated in Table 3, the super-conducting material for UWMAK is copper-stabilized NbTi, and for the PPPL design, Nb_3Sn.

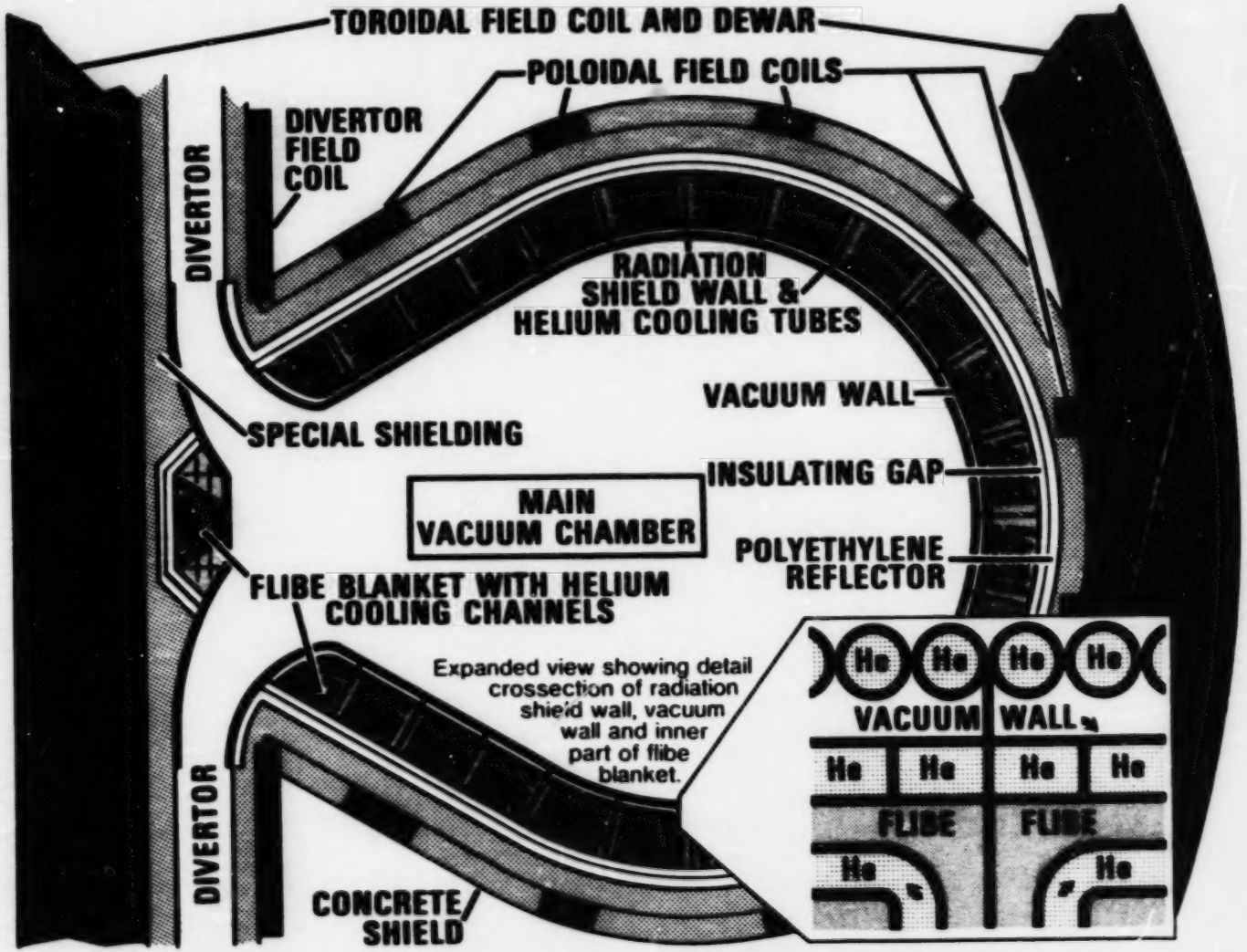

Figure 6. Princeton Tokamak Cross Section Showing Blanket, Shielding, and Main Field Coils

TABLE 6. MAGNET CHARACTERISTICS

	UWMAK	PPPL
Toroidal Field Magnet System		
Number of Coils	12	48
Magnet - Coil WT (MT) each	900	130
Max Magnetic Field Flux Density (gauss)	86,600	160,000
Flux Density at Plasma Center (gauss)	38,200	60,000
Stored Energy in Coils (Joules--Total)	350×10^9	250×10^9
Coolant	Liquid He Bath	Forced Supercritical He Vapor
Max Magnet Structure Design Stress (psi)	90,000	N. A.*
Poloidal Field System		
	Diverter Coils (4 pr)	Diverter Coils (2 pr)
	Transformer Coils (5 pr)(to supply plasma current)	Vertical Field Coils (5 pr)(to heat and ignite plasma)
		Control Field Coils (21 pr)(to maintain plasma current)

* Not available.

The size of the toroidal magnet-coil systems required impose a number of design problems. These magnets are incorporated into the modular segment design of the reactor blankets. The bulk and mass of the coils pose problems in fabrication, handling, and maintenance. Mechanical forces induced by the strong magnetic fields on coil conductors and support structures require careful design to keep stresses below yield points.

Cooling System and Steam Generators

The three-loop (Li-Na-steam) cooling system for the Wisconsin design is shown in Figure 7. This design is used primarily to minimize tritium and corrosion product transport from the lithium primary to the steam generators. There are 12 three-loop systems, including steam generators, which feed two 1,800 RPM tandem compound turbine generators. Condenser cooling is provided by two conventional units rejecting heat to the atmosphere via mechanical draft-cooling towers.

Power distribution and coolant loop operating parameters for the Wisconsin reactor are summarized in Table 7. In addition to the main coolant system, one three-loop system is used to cool the divertor coils. This is a relatively low temperature system producing steam at 188°C, which is used to drive the main steam generator feed water heaters. A third cooling system handles energy removal from the main blanket shield, which must be maintained well below the melting temperature of lead.

The PPPL cooling system consists of a helium primary and 12 helium-driven steam generators designed especially for this application. These high-temperature, gas-cooled fission reactors (HTGR) represent state-of-the-art technology. Each steam generator, a super-critical, double reheat unit with a base-mounted, axial flow helium circulator, operates in parallel with the main helium circuit. Helium circulates through the reactor at 55 million kg/hr.

An important part of the reactor coolant system in the UWMAK is the tritium-handling system associated with the tritium-breeding main coolant. Tritium breeding in the PPPL does not occur in the main coolant, but in the flibe blanket. These handling and clean-up systems will be discussed in the section on radioactive waste systems.

SOME COMPARISONS OF FUSION AND LIGHT WATER FISSION REACTOR CHARACTERISTICS

The core of a light water reactor is a heterogeneous mixture of fuel, cladding, structural material, and water; whereas, the core or energy-producing region of a fusion reactor is a dilute mixture of fuel atoms in the plasma state. Control is achieved in a fission reactor by regulating the number of neutrons available to initiate fission reactions. Fusion reactor control (or, more

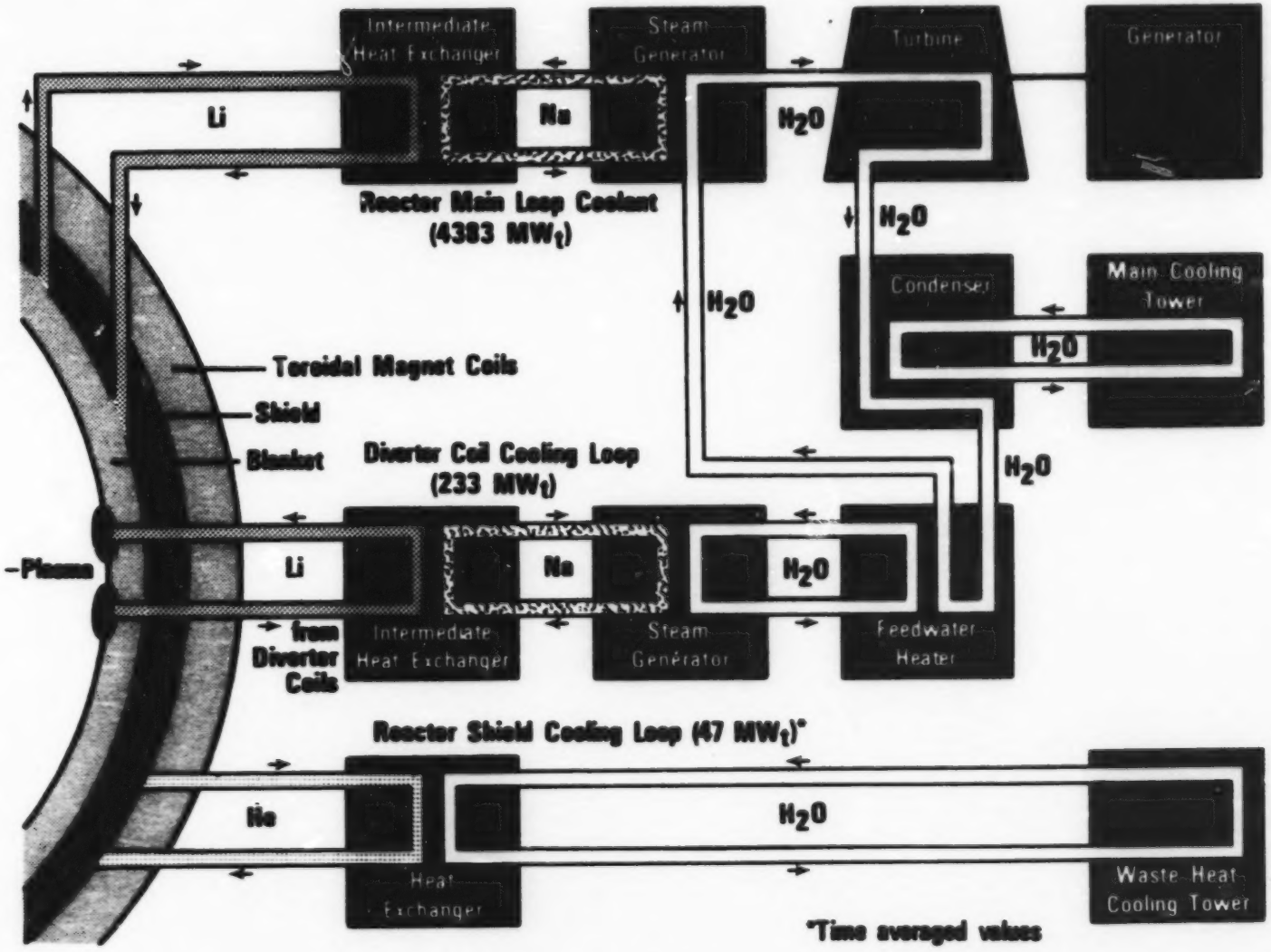

Figure 7. Wisconsin Toroidal Fusion Reactor Coolant Loops

TABLE 7. WISCONSIN TOKAMAK COOLING
SYSTEM POWER DISTRIBUTION AND OPERATING PARAMETERS

SYSTEM	NUMBER OF LOOPS/UNITS	Mw-th EACH	MASS FLOW IN EACH LOOP (Kg/hr)	LOOP TEMP HOT/COLD (°C)	OPERATING PRESSURE (psi)
Blanket:					
Li Primary	12	392	2.7×10^6	489/359	670
Na Secondary	12	392	9.8×10^6	456/336	35
Steam	12	392	6.3×10^3	404/218	2,000
Diverter:					
Li Primary	1	250	3.3×10^4	325/	20
Na Secondary	1	250			
Steam	1	250	3.2×10^5	188/57	N. A.*
Shield:					
He Coolant	12	4.2	1.9×10^4	200/50	735
H_2O	12	4.2	2.9×10^5	N. A.*	N. A.*
Turbine Generator:					
Turbine Generator	2	840 MWe	3.79×10^6 (steam)	399/218	1,900
Condenser	2	1,450	9.4×10^7 (Cooling H_2O) (4.1×10^5 gpm)	13.3 (ΔT)	1.72 (Back pressure)

* Not available.

accurately, reaction rate maintenance) is essentially accomplished by regulating the rate at which fuel atoms are supplied to the plasma.

Thus, in contrast to a fission reactor core with a fuel inventory of around 100 metric tons (MT) of uranium, a fusion reactor plasma contains only a gram or two of fuel at any time and fuel is supplied continuously to maintain the reaction rate. Nuclear energy in a fission reactor is primarily carried by the fission recoil atoms and converted to heat within a few millimeters of the individual reaction sites in the fuel pins. On the other hand, fusion energy is primarily carried by high-energy neutrons (about 80 percent of the fusion energy) and converted into thermal energy in the reactor blanket within several meters of the plasma or energy-producing zone.

A rough analogy can be made between the core inside the pressure vessel of a fission reactor and the plasma volume of a fusion reactor. The primary heat transfer surfaces are the fuel pin cladding and vacuum wall, respectively. In addition, the cladding is the primary container for the reactor fuel; whereas, in the fusion device, the primary container is the magnetic field surrounding the plasma. The power density in a light water reactor is around 100 kw/liter of core volume; in the conceptual fusion plant designs, it is only one or two kw/liter of plasma volume. Hence, the much larger reactor size is required for a fusion plant of comparable power output. The analogy between the cladding and vacuum wall as primary heat transfer surfaces suffers in that the energy flux through the cladding of a fission reactor is mostly thermal energy; whereas the "wall loading", as it is called in a fusion reactor, is primarily neutron kinetic energy with relatively little thermal energy transfer to the wall itself. In the Princeton design, roughly 13 percent of the total thermal energy produced is carried by alpha particles from the plasma and the majority of this fraction is deposited as heat in the vacuum wall (Mills, 1974).

Fuel temperatures in fusion reactors must approach 50 to 100 million degrees centrigrade. In LWR's, maximum fuel rod center design temperatures are around 1,900°C. A major difference in thermal characteristics between the two systems is that the plasma fuel has essentially zero heat capacity while the UO_2 fuel in a light water reactor has a high heat capacity. The result of this, and the fact that the UO_2 fuel contains heat-producing fission products, is that a LWR core has tremendous thermal inertia and a fusion core or plasma has very little. Emergency core cooling, per se, is not a problem for a fusion reactor. Maximum LWR cladding and fusion first wall design temperatures are comparable as they are both constrained by material properties.

Operating pressure in a pressurized water fission reactor (PWR) core is in the neighborhood of 2,000 psi and results from thermal-hydraulic requirements. Pressures at the vacuum wall inner surface in a fusion reactor are very low, on the order of several microns

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of Hg. The pressure of the plasma itself is in the range of ten atmospheres, but is contained by the magnetic fields within the vacuum chamber.

The radiation environment in a fusion reactor is expected to be more severe than in fission reactors. Neutron flux densities in the Wisconsin and Princeton fusion designs, for example, are several times greater than in an LWR. In addition, the average neutron energy approaches 14 MeV in fusion plasma environs which produces more radiation damage than neutrons of fission energies.

Table 8 summarizes a number of comparisons between UWMAK-I (Wisconsin, 1974), the PPPL design (Mills, 1974), and the Combustion Engineering System 80 PWR (Combustion Engineering Company). Further comparisons will be made.

RADIOACTIVE WASTE MANAGEMENT SYSTEMS

Fusion power plants, like fission plants, can be expected to produce gaseous, liquid, and solid radioactive wastes. While fusion plants will not produce fission products and actinides, tritium and a variety of activation products will constitute the sources of radioactive waste. The majority of wastes associated with routine operation will be produced in the various tritium-handling and corrosion product removal systems in the blanket and plasma exhaust clean-up systems.

An idea of the magnitude of the tritium-handling problem can be gained from an examination of Figures 5 and 8. Figure 5 shows the principal fuel flows in the PPPL design and Figure 8 shows the tritium flows and environmental release sources for the Wisconsin design. For example, in the Wisconsin design, there are a minimum of 36 blanket and shield coolant loops from which tritium is recovered. In addition, there are tritium recovery systems in the vacuum systems and in the divertor system cooling loop. Characteristics of the tritium extraction systems are given in Table 9 for the Wisconsin design.

Another major source of radioactive waste will be activated corrosion products in the various coolant loops, since these regions lie within the zone of significant neutron fluxes in the blanket. Details of treatment and handling systems and procedures for this category of waste are not currently available in the designs under consideration. The Wisconsin design analysis includes a discussion of corrosion product formation in the coolant and the neutronic analysis of activation product formation in stainless steel structural materials. What is currently lacking is the coupled analysis to predict coolant system radioactivity inventories which are necessary to determine coolant-radioactivity clean-up system design requirements. It is expected that liquid metal cooled fast breeder reactor (LMFBR) design experience will be of value in this area.

TABLE 8. COMPARISON OF SOME FUSION AND LIGHT WATER (PWR) REACTOR CHARACTERISTICS

PARAMETER	CONCEPTUAL FUSION REACTORS		COMBUSTION ENGINEERING SYSTEM 80 PWR
	UWMAK-I	PPPL	
Power (Mwt)	4,600	5,305	3,800
Plasma/Core Vol. (m^3)	6,400	2,190	40
Plasma/Core Fuel Inventory (Kg)	2.1×10^{-3}	0.9×10^{-3}	1.03×10^{5}
Power Density (Kw/l)	0.8	2.4	95.6
Heat Transfer Surface Area (m^2) First Wall/ Cladding	2,830	1,563	6,410
Average Neutron Loading/ Heat Flux at First Wall/ Cladding (watt/cm^2)	177	211	57
Max Fuel Temp. (°C)	1.1×10^{8}	3×10^{8}	1,880
Max First Wall/Cladding Temp (°C)	470	680	350
Neutron Flux Density - All Energies (n/cm^2-sec)	N. A.*	8.7×10^{14} (at wall)	2.9×10^{14} (core center)

* Not available.

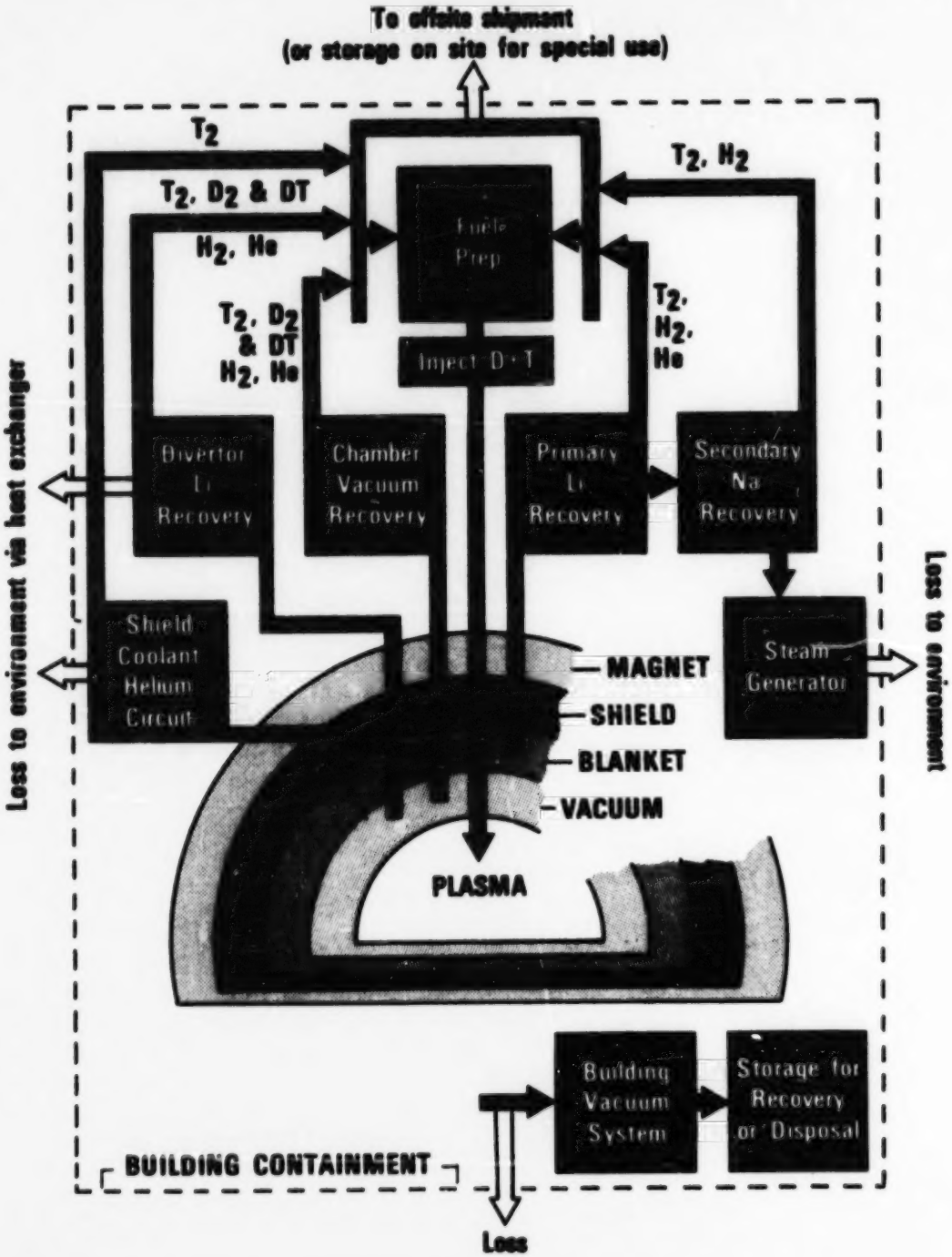

Figure 8. Tritium Flow in Wisconsin Tokamak

TABLE 9. SUMMARY OF UWMAK-I
TRITIUM EXTRACTION SYSTEM CHARACTERISTICS
(WISCONSIN, 1974)*

COOLANT SYSTEM	TEMP. RANGE °C	EXTRACTION METHOD	TRITIUM ACCUMULATION PER DAY (kg)	TRITIUM LEAKAGE Ci/DAY	TOTAL Na OR Li (kg)	TRITIUM CONCENTRATION IN Li OR Na ppm (wt.)	TRITIUM INVENTORY (kg)
Primary Lithium	283-489	Yttrium Metal Bed	1.05**	10.1	1.73×10^6	5	In Li 8.7 In Beds 1.0
Secondary Sodium	261-411	Yttrium Metal Bed	~0		7.6×10^5	8.7×10^{-5}	In Na 6.7×10^{-5} In Beds ~0
Diverter Lithium Sodium	200-325 190-265	Yttrium Metal Bed	7.4 T + 5.0 D	2×10^{-4}	3.4×10^4 6×10^4	0.24 3×10^{-4}	In Li $8 \times 10^{-}$ In Beds 3.5 In Na $6 \times 10^{-}$
Diverter Vacuum	25	Charcoal-Cooled with Liquid He	0.3 T + 0.2 D	1×10^{-4}		Not Applicable	0.3
Shield Helium	50-200	Metal Getter	1.1×10^{-6}	low		Not Applicable	low
			Total:	10.1			13.5

* Based upon thermodynamic calculations; no kinetic considerations.
** At maximum breeding ratio of 1.49

Discussions of solid waste treatment systems are limited to general discussions of sources of activated components, such as vacuum and divertor wall replacement, failed pumps and equipment, and solids or sludges from the various tritium and corrosion product clean-up systems. Estimates are made of amounts of solid waste expected and several modes of handling are discussed.

Gaseous wastes are expected from various tritium leakage paths, and neutron activation of air and coolant gas contaminants. Both conceptual designs include the use of containment-building vapor barriers around the main torus to control gaseous wastes. In fact, the Wisconsin design includes an evacuated primary containment building to limit tritium out leakage. No details were provided of general plant gaseous waste clean-up and treatment systems, including building ventilation systems. No consideration was included for secondary sources of gaseous wastes, such as cooling system off gassing, and turbine condenser air ejectors.*

A similar situation exists with respect to liquid waste treatment systems. The balance of plant designs do not include integrated liquid waste treatment systems to handle the various expected sources of liquid radioactive wastes.

In summary, detailed assessment of the environmental radiological consequences of fusion power plants must await the arrival of complete power plant system design details. Until such details are available, plant pathways and environmental release points cannot be identified and environmental release quantities estimated. It is obviously unreasonable to expect to have the design details of radioactive waste systems in hand, as the current conceptual designs are at a relatively early stage of evolution toward actual plant designs. The discussion of environmental radiological impacts that follows will therefore be limited by the amount of design information available.

Based on the examination conducted herein of sources of radioactive materials produced in these fusion power plant designs, it can be seen that radioactive waste treatment will require the development of major in-plant systems. It is expected that these systems may equal or exceed in cost and complexity the radwaste systems in light water nuclear plants of comparable size.

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* Several hundred moles of gases are produced each day in the Princeton flibe blanket by neutron transmutations. The bulk of this is He, but significant amounts of N-16, O-19, F-18, and F-20 are produced. With the exception of F-18, all have half lives of less than 30 seconds. The equilibrium inventory of 110 minute F-18, however, is 6.6×10^8 Ci.

SAFETY ASPECTS

As in any engineering design effort, safety considerations form an inherent part of the design process. An important benefit of the current conceptual design exercises for fusion power plants is the early identification of safety and environmental constraints that must be factored into the design iteration.

The safety design objectives for fusion power systems are generally the same as for any nuclear power system: to design a system that can be built and operated in a manner that poses risks to the public that are within acceptable limits commensurate with the benefits gained. The primary risk to the public is expected to be from the exposure to radioactive materials released from the facilities.

Some safety aspects are dictated by the nature of the fusion process itself, while others are highly dependent upon the design of a particular system (Fraas, 1974). For example, from the conceptual designs examined in the previous section, a number of safety requirements can be identified, and design requirements for safety systems can then be considered for reactors with these characteristics. It is equally important to identify and specify appropriate design safety criteria for fusion power reactors, e.g., design basis accidents, or probabilistic-type criteria, in a timely manner as designs evolve toward actual systems.

In the following discussion, the main emphasis will be on safety aspects of tokamak systems represented by the Wisconsin and Princeton designs.

DESIGN FOR SAFETY

For fusion reactors designed around the D-T reaction, the major safety design objective is to achieve a design which will minimize the release of tritium and other radioactive materials to the environment. Additional potential hazards to members of the public may be posed by large inventories of materials, such as lithium, mercury, lead, boron, and in some designs, beryllium (Kastenberg, 1975).

A brief discussion of some of the safety considerations associated with the various systems of the tokamaks follows. They will be considered under the general headings of reactor, coolant system-blanket, and structure and balance of plant. The emphasis

here is on public safety considerations, but it is noted that the design for plant personnel safety also poses a challenge. The problems of worker protection against radiation exposure from massive tritium inventories and direct radiation from intense 14 MeV neutron and associated radiation fields will pose challenges to both designers and operational health physicists. In addition, electrical hazards will be present and intense magnetic fields over large areas will raise questions concerning safety and biological effects. Liquid metal coolants pose in-plant safety problems and a variety of toxic materials associated with various systems will be present.

Overall Reactor Design

An important safety design problem associated with fission reactors is not present in fusion devices. Fission reactor cores are designed around the problem of avoiding a nuclear excursion or excessive reaction rate leading to uncontrollable energy release. The problem in a plasma fusion device is to design a system that will maintain the delicate balance of conditions required to sustain the reaction. The unlikely instantaneous reaction of the entire plasma fuel inventory would be expected to result in only a minor increase in blanket temperature (Fraas, 1974).*

A major design objective for a power plant is to "minimize the total inventory and the loss rate of tritium in every sector of the plant" (Mills, 1974).

In the Princeton design, these objectives will be pursued by (Mills, 1974):

(a) processing the blanket so as to achieve the lowest possible concentration of tritium,
(b) employing a contiguous system of barriers (both metal and ceramic) that are impervious to the hydrogen isotopes in any form, and
(c) operating a processible gaseous purge flow system around and throughout the plant on a continuous basis.

Coolant System--Blanket

Two important goals of fusion reactor coolant system-blanket design are: 1) maximum integrity, and 2) minimum radioactivity inventory. Meeting these goals and satisfying other design constraints such as cost, high temperature operation for maximum thermodynamic efficiency, resistance to radiation damage, availability, and fabricability, are serious challenges.

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* This assumes that the burn occurs uniformly throughout the plasma volume. The possibility of non-uniform plasma energy overshoots leading to locally high energy deposition and vacuum wall damage is being investigated (Kastenberg, 1975).

The need for maximum integrity is obvious, i.e., to reduce the leakage of tritium and other radioactive materials under normal and abnormal operating conditions. Minimization of the radioactivity inventory from neutron activation of materials is desirable to reduce the decay afterheat removal problem, to make the source of radioactive materials for release to the environment as low as practicable, and to minimize solid radioactive waste handling and disposal problems.

Preliminary comparisons of afterheat and induced radioactivity have been made between various fusion reference designs and with LWR and LMFBR fission reactors (Dudziak, 1975; Conn, 1975). Comparisons of fusion blanket operating average power densities with fission reactor core power densities indicate that fusion reactors range from 5 to 20 Mwt/m^3 of blanket volume; LWR's are around 100 Mwt/m^3; and LMFBR designs range from about 300 to 600 Mwt/m^3. Initial shutdown afterheat to operating power ratios for several years' operation are in the range of 0.5 to 5 percent (depending upon blanket design and operating history) for the various fusion designs and about seven or eight percent for a fission reactor.

The post-shutdown decay afterheat removal problem appears to be much less severe in fusion reactors than for most fission reactors. More important than the fact that initial decay heat ratios are slightly lower is that the relevant afterheat power densities are lower by a factor of 50 to 100.

In terms of the radioactive materials present and potentially available for release to the environment as a result of a disruptive accident, fusion reactors at this stage look favorable in comparison with fission reactors. There are no fission products or actinides present, of course, and the major species are isotopes of Fe, Ni, Mn, Co, Cr, V, and Nb, depending upon the blanket materials, and tritium. Additionally, relatively short-lived isotopes will be produced in blanket coolants and breeding materials, e.g., isotopes of He, Li, N, O, and F, in flibe, and Na-22 and Na-24 in sodium.

In accident analyses of large power systems, stored energy in various systems and the potential for its uncontrolled release must be examined. Major energy sources in the fusion plants are: the liquid metal or molten salt and gaseous coolants, the super-conducting magnets, the kinetic energy of the plasma, and the energy stored in cryogenic systems, e.g., liquified helium. Stored energy in coolant for the fusion designs under consideration appears to be comparable to an LMFBR, i.e., on the order of 10^{12} Joules (Postma, 1971). Unique to fusion reactors is the large amount of stored magnetic energy in the super-conducting magnets, between 10^{11} and 10^{12} Joules, for the Wisconsin and Princeton Tokamaks.

Structure and Balance of Plant

Several features can be identified in the fusion designs under

examination, which have a bearing on safety. The structural design must accomodate the vary large weights of the magnets, blanket shield, and supports. The modular blanket designs require that the segments be transportable and capable of precision placement. For example, each module of the UWMAK weighs over 3,000 MT, which far exceeds conventional crane capabilities and for which special transport systems will have to be designed. The removal and replacement of blanket sections may have to be handled remotely due to high radiation levels.

Both the Wisconsin and Princeton designs include containment vapor barriers surrounding the reactors. The Wisconsin design includes an evacuated enclosure surrounding the nuclear island and an additional outer containment building, intended to minimize tritium leakage to the environment.

The Wisconsin three-loop cooling system design with the intermediate sodium loop is primarily intended to reduce tritium and corrosion product leakage from the primary lithium loop into the steam loop. The use of 12 of these three-loop cooling systems is also motivated by the desire to limit radioactive material losses in the event of a catastrophic, disruptive accident involving the primary lithium coolant system.

POTENTIAL ACCIDENTS AND ENGINEERED SAFETY FEATURES

Both the Wisconsin and Princeton design reports include an analysis of potential accidents. An independent study of safety aspects of tokamaks centering on the Wisconsin design is currently underway (Kastenberg, 1975). What follows is a brief summary of the accidents considered and some preliminary conclusions concerning safety of these designs.

A number of events have been identified which could lead to major release of tritium to the environment. Under general categories, these include: loss of coolant and coolant flow; liquid metal fires and chemical reactions; plasma overpower; magnet failures; heat exchanger and steam generator failures; liquid helium warmup; vacuum system, including first wall failure; missile generating accidents; and tritium handling accidents. External initiating events should also be considered, such as earthquakes, floods, tornadoes, aircraft collisions, and sabotage.

The Princeton and Wisconsin design studies contain a discussion of potential accidents postulated for each of the major plant systems. A design basis accident for the Princeton design has been identified. This is considered to be the accident with the greatest public health consequences, i.e., the accident resulting in the greatest release of volatile radioactive material. The postulated design basis accident is the release of the entire primary fuel loop tritium inventory, about 2.5 kg, to the reactor building.

As a result of this analysis, an engineered safety system has been identified. This consists of a building containment system, similar to those associated with LWR designs. Additional systems, such as containment atmosphere clean-up systems, were also discussed.

Probabilistic safety assessments leading to absolute public risks associated with fusion reactor operations are not feasible at this time. These methods generally require complete system design details and knowledge of operating modes (Hafele, 1974). An evolution in safety assessments beyond the identification of postulated accident outcomes is currently being pursued. These efforts include the examination of event sequences and failure modes which can lead to various accident end events (Wisconsin, 1974; Kastenberg, 1975).

ENVIRONMENTAL IMPACTS OF NORMAL OPERATION

RADIOLOGICAL EFFLUENTS

Tritium, primarily, and additional amounts of activation products are expected to constitute the main radiological effluents. From routine operations, it is expected that both gaseous and liquid releases will occur. While expected release rates are unavailable, some discussion has occurred as to what practicable design tritium release rates might be. Inventories of various isotopes estimated for Wisconsin and Princeton Tokamak designs are summarized in Table 10. In comparison with light water reactors, the estimated tritium release rates for these fusion designs are about one to two orders of magnitude higher. To achieve this level of release requires that leak rates be one to two orders of magnitude lower than currently achieved in light water reactors. For example, a boiling water reactor (BWR) leaking about 100 Ci of tritium per year corresponds to a leakage fraction of about 10^{-5} of the reactor coolant inventory per day (Pigford, 1973).*

Gaseous effluents are expected from routine operation as indicated previously. No estimates of release rates are currently available for the designs under consideration. Candidates for potential release include the large inventory of F-18 (7×10^{8} Ci) evolved at a rate of about 70 moles per day of fluorine from the flibe blanket in the Princeton Tokamak. Fluorine is very reactive chemically, and it is expected that most of the fluorine evolved will appear in the flibe as various compounds, such as fluorides, some of which may be gases. It is apparent that there will be

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* Tritium diffusion rates through some fusion reactor candidate materials are not well known. Preliminary data which is becoming available suggests that lower leak rates than previously estimated for the designs under discussion may be attainable (Young, 1976).

TABLE 10. SUMMARY OF RADIOLOGICAL QUANTITIES ASSOCIATED WITH TOKAMAK DESIGNS

	UWMAK	PPPL
Tritium Inventory (kg)	13.5[a]	2.6
Estimated Tritium Environmental Release Rate (Ci/Day)	10	6
Fractional Tritium Leakage of Total Inventory Per Day	0.7×10^{-7}	2.3×10^{-7}
Major Isotope Inventories in Reactor Blankets (Ci)	Mn-56 (2.0×10^{8})[b] Fe-55 (6.5×10^{8}) Co-58 (1.4×10^{8}) Mn-54 (1.1×10^{8}) Co-60 (2.2×10^{7}) V-49 (3.1×10^{6}) Ni-57 (5.1×10^{6})	F-18 (6.6×10^{8})[c] Fe-55 (1.5×10^{9}) Co-57 (2.5×10^{9}) Mn-54 (1.5×10^{8}) Co-60 (1.5×10^{8}) V-49 (1.7×10^{9})
Estimated Annual Quantities of Solid Waste (m^3/yr)		
First Wall Replacement	31.4	20.7 (from first wall and blanket
Annual Equivalent from Replacement of Blanket	62.6	
Corrosion Products Removed from Coolant	0.3	45 (from flibe and structural material)

(a) Estimates of tritium inventory for UWMAK vary, other reports indicate higher values than the design report; 15-30 kg (Conn, 1974) and 30-50 kg (Sehnert, 1975) respectively.
(b) First wall inventory only, based on 10-year operation.
(c) Entire blanket inventory calculated for 30-year operating history.

large potential sources for tritium release as HTO vapor and T_2 and D-T gas. Another source of gaseous effluent is A-41 from neutron activation of air and cover or off gases in various liquid metal systems. In the Princeton design, a major potential source of this isotope is the several hundred grams per hour of argon that is circulated through the plasma as an impurity for burn control and quenching.

The question of exposure from direct and scattered radiations originating from the fusion reactor itself, to members of the public in the environs, has not been analyzed in the design studies.

Potential sources are: the large 14 MeV neutron fluxes, neutron capture gammas, and gammas from neutron activation products. In the Princeton design, while no numerical estimates were given, it appears that the equilibrium inventory of N-16 in the flibe is on the order of megacuries. Depending on the amount of shielding around the flibe, this could constitute a source of exposure, i.e., the air-scattered secondaries from the energetic N-16 decay gammas. For example, nearest boundary, off-site annual dose rates for the TFTR from direct radiation have been estimated to be between 3 and 5 millirem per year (ERDA, 1975a). This is based on about 300 equivalent seconds of operation at 15 MWt plasma fusion power.

Sources of solid radioactive wastes from routine reactor operation include: sludges from various coolant clean-up systems, activated materials and components from wall and blanket replacement and failed equipment, and miscellaneous contaminated small tools and clean-up wastes.

In the Wisconsin design, an estimated 1,500 to 2,000 kilograms of stainless steel is expected to be dissolved in the lithium coolant per year (Conn, 1974). This material will have to be removed continuously to avoid fouling of heat transfer surfaces. An additional source of radioactive materials in coolants is from neutron sputtering of first wall materials into the coolants. It has been estimated that this mechanism could contribute up to 50 percent as much radioactivity as chemical corrosion alone (Conn, 1974).

Equilibrium Li coolant corrosion product inventory total radioactivity is estimated to be about 5×10^7 Ci (Wisconsin, 1975). This indicates that the specific activity of coolant-borne corrosion products will be on the order of curies per gram. These materials from coolant clean-up systems will have to be handled as high-level radioactive waste requiring biological shielding, and perhaps cooling as well.

A major source of solid waste generated during routine operation is the activated structural material from periodic replacement of first walls and blankets. Some volume estimates obtained from the Wisconsin and Princeton design reports are included in Table 10. This material will also have a specific activity on the order of curies per gram, particularly from first wall material (Wisconsin,

1975). The solid waste handling problems posed by these materials are unique to fusion designs in that large volumes of high specific activity solid wastes are not associated with fission reactor routine operation. The conceptual fusion designs include provisions for remote handling of this material via manipulators and hot cells. It is likely that the volumes of this material that will actually have to be handled have been underestimated in the design reports, as it does not appear to be amenable to compaction. The importance of minimizing blanket and structural radioactivity has been recognized by fusion reactor designers and it is expected that future designs will improve on the Princeton and Wisconsin designs in this regard.

While fusion reactors do not require off-site shipment of highly radioactive spent fuel as with fission reactors, it appears that shipments of high-level radioactive waste will be required unless it is decided to locate fusion plants such that on-site, long-term storage of these wastes is feasible.

It can also be expected that an additional several thousand cubic feet per year of solid radioactive waste will be generated from miscellaneous sources as in any large nuclear power plant. These could include: filters from the heating and ventilation system, deactivated coolant traps as in an LMFBR (AEC, 1974) analytical laboratory and liquid waste treatment residues, contaminated small tools and parts, clean-up wastes such as towel wipes and plastic bags, and protective clothing. Additional materials unique to fusion plants would include: vacuum system traps, tritium clean-up system beds, tritium getters, tritided metals used to contain tritium fuel, and failed fuel system components. Decommissioning of fusion power plants at the end of design lifetime would be expected to present waste disposal problems similar to those associated with LWR power plant decommissioning.

NON-RADIOLOGICAL IMPACTS AND SITING REQUIREMENTS

The fusion designs examined herein have conventional steam power cycles with heat rejection to the atmosphere via evaporative cooling towers. It can be expected that these plants would have chemical discharges similar to conventional nuclear or fossil plants of similar size. The thermal impacts could be estimated by scaling the various material balances for an evaporation cooling system to the rejected heat quantities for these plants.

Siting requirements for the fusion designs, at this stage, appear to be similar to those for conventional large LWR's. In terms of non-radiological requirements, few differences are expected from any large fossil or LWR plant. Designs have not evolved to the stage where accident-based siting criteria have been developed. These designs appear to have radiological implications quite similar to LWR's, particularly from routine operations. It is not reasonable to expect that these designs will have much less stringent siting requirements than LWR's.

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